Industry Response to the NRC's Comments (September 18, 2001 letter) on Rev. 6 Draft of SG Examination G/L and Staff's Request for Additional Information

(NRC Comments and Questions are in *Italic*)

Preliminary Conclusions

The staff believes that the industry's prescriptive proposal should not significantly increase risk subject to certain additional provisions identified below. This finding is contingent on receipt of additional information from industry indicating a clear preponderance of evidence that early experience with Alloy 600 TT and 690 TT tubing and sleeves has been crack free. That said, the staff recommends that the industry revise its proposal to incorporate these provisions which are as follows:

1. *The one fuel cycle limitation should be one fuel cycle or 24 EFPM, whichever is shorter. Similarly, the two cycle limitation should not exceed 48 EFPM and the three cycle limitation should not exceed 72 EFPM.*

   Section 3 in Rev.6 of the PWR SG Examination Guidelines removes the reference to "skipping" fuel cycles and establishes the limits of 24 EFPM for 600MA, 48 EFPM for 600TT, and 72 EFPM for 690TT as the maximum length of time that a SG can operate without being inspected.

2. *Definition of "active damage mechanism" should be redefined as follows:*

   Active ~~damage mechanism~~ degradation:

   - *A combination of ten or more new indications of degradation (>=20% TW) and previous indications of degradation which display an adjusted, average growth rate equal to or greater than 25% of the repair limit per ~~cycle~~ inspection interval in any one SG. Adjusted growth rate refers to scaling the growth rate for the previous inspection interval to reflect the length of the next scheduled inspection interval. For example, if the next schedule inspection interval is twice the length of the previous interval, the adjusted growth rate is twice the value observed over the previous inspection interval.*

   - *one or more new or previously identified indications of degradation~~, including cracks,~~ which display ~~a~~ an adjusted growth rate greater than or equal to the repair limit in ~~one cycle of operation~~ per inspection interval.*

- *damage related to loose parts or foreign objects is subject to the above criteria, irrespective of whether the causal objects are believed to have been retrieved.*

- *any indications associated with cracks*

Active Damage Mechanism is defined in Appendix F, Rev. 6 of PWR SG Examination Guidelines as:

- **A combination of ten or more, new indications (≥ 20% TW) of thinning, pitting, wear (excluding loose part wear) or impingement and previous indications which display an average growth rate equal to or greater than 25% of the repair limit in one inspection-to-inspection interval in any one SG,**

- **One or more new or previously identified indications (≥ 20% TW) which display a growth greater than or equal to the repair limit in one inspection-to-inspection interval, or**

- **Any crack indication (outside diameter IGA/SCC or primary side SCC).**

The nature of loose parts does not fit the definition of Active Degradation Mechanism. The actions required upon the identification of loose part degradation are not the same as those that would be pursued in response to other forms of degradation. An evaluation is required that addresses programmatic and inspection limitations as well as the specifics of the actual condition. The results of this evaluation shall be considered in the degradation, condition monitoring, and the operational assessment. Loose parts requirements are contained in Sections 3.8 and 6.10.3 of the PWR SG Examination Guidelines.

Industry agrees that growth rate should be defined on an inspection-to inspection-interval as defined in the above definition of Active Damage Mechanism. However, adjusted growth rates are not part of the PWR SG Examination guidelines. The application of growth rate (including growth rate adjustment) in defining acceptable operating interval is addressed in the EPRI SG Integrity Assessment Guidelines. The Tube Integrity Ad-hoc committee will consider this comment.

3. *For plants with Alloy 690 TT tubing, three cycle inspection intervals shall be preceded by a two cycle inspection interval.*

   Rev. 6 of the PWR SG Examination Guidelines requires that 100% of tubes be inspected at pre-service and the end of the first fuel cycle. Thereafter multiple cycle inspection intervals are allowed provided that the 72 EFPM limitation is not exceeded and that the inspection interval is supported by an operational assessment.

   Preceding a three cycle interval by a two cycle interval is not necessary. Tubes are in their best condition early in life. Operating history of SGs with alloy 600TT and 690TT tubes indicates that any problems that may eventually occur do not exhibit themselves until well after three cycles.

   Justification for the inspection intervals is provided in the answer to Requested Information item 1 below and the attached report, "Experience of U.S. and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves".

4. *The initial finding (industry wide) of indications associated with a cracking mechanism shall define the "time to detectable cracking threshold" for Alloy 600 TT SGs or Alloy 690 TT, as applicable. The time to cracking threshold shall be normalized to a reference temperature. The licensee shall take action as necessary to ensure that cognizant personnel at all plants utilizing the same tubing material are promptly informed of the finding. Upon receipt of such information, the other licensees shall consider the information as part of the degradation assessment which is to be performed prior to the next scheduled refueling outage to assess the need for modification to the schedule for the next SG inspection. Inspections shall be performed at each refueling outage after the equivalent accumulated full power operating time on the SGs (i.e., normalized for reference temperature) exceeds 75% of the "time to detectable cracking threshold."*

   The use of a "time to detectable cracking threshold" tied to one plant's experience does not take into account the unique nature of each SG's operating conditions. The industry's proposed inspection periods are selected to be sufficiently conservative for generic application based on the current knowledge of 600TT and 690 TT materials (see attached report, "Experience of U.S. and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves").

   The industry SG Program guidelines address potential degradation mechanisms through the degradation assessment. The requirements for degradation assessment have been expanded in revision 6 (see section 5.2). The degradation assessment required at each refueling outage must include consideration of

industry experience with regard to any newly found form of degradation and its impact on the upcoming condition monitoring campaign. Industry experience is captured by item 25 of Section 2.3 in the PWR Steam Generator Examination Guidelines, which requires, when requested, communication of examination results to other plants, vendors, and EPRI. In addition, item 30 of Section 2.3 requires documentation of completed examination and results in the EPRI Steam Generator Degradation Database within 120 days. This database is maintained on an EPRI web site and is accessible to utilities and their inspection vendors. Additionally, plant experience reports are shared no less than three times per year among utilities attending the technical Advisory Group meetings of the EPRI SGMP.

Finally, it should be noted that per Section 4.1, NRC approval is required prior to the implementation of performance based inspection intervals.

Therefore, time-to-cracking threshold and its temperature-normalized value for each plant is not considered, by itself, to be a critical detail to be mandated in the inspection interval requirements of Chapter 3.

5. *The "time to detectable cracking" should be revised downward as necessary to lower bound subsequent findings (industry wide) of crack indications occurring after equivalent, accumulated full power operating times less than that observed earlier. Again, the affected licensee shall take action as necessary to ensure that cognizant personnel at all plants utilizing the same tubing material are promptly informed of the finding. The other licensees shall respond as described in item 4.*

   See response to Item 4.

6 *For purposes of tube integrity assessments supporting multi-cycle inspection intervals, ligament tearing of volumetric flaws shall be considered "burst." That is, volumetric flaws should have a factor of three margin against such ligament tearing.*

   Industry disagrees with the position taken by Staff, and regards the information to be a change to a previously agreed technical position. The basis for this conclusion is as follows.

   1. **A meeting was held at the NRC Offices on July 24, 1999. An agenda item identified by the NRC project manager, Tim Reed, was final approval of the industry white paper on the "Definition of Burst". At that meeting, the white paper was presented and discussed. The NRC Staff concluded at the time that the definition was acceptable and considered the item closed. The definition was subsequently incorporated into NEI 97-06 and the white paper was included in the EPRI *Steam Generator Integrity Assessment Guideline*.**

With respect to ligament "pop-through" the following information was contained in the white paper

"The definition (burst) is also intended not to characterize local instability, or for example, "ligament pop-through", as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. As an example of not having a burst, consider an axial crack about 0.5" long with a uniform depth at 98% of the tube wall. Deformation during pressurization would be expected to lead to failure of the remaining ligament, (i.e., extension of the crack tip in the radial direction) at a pressure below that required to cause extension at the tips in the axial direction. Thus, this would represent a leakage situation as opposed to a burst situation and a factor of safety of three against crack extension in the axial direction may still be demonstrated. Similar conditions have been observed for deep wear indications."

Consequently the interpretation presented in the paper titled *Structural Integrity of SG Tubes with AVB Wear*, by R. Keating (W), H. Lagally (W), and R. Lieder (Seabrook), June 2001 is consistent with the agreed to burst definition for the type of AVB flaws found in Westinghouse Model F steam generators. The position in this paper is considered defect specific and would not necessarily apply to other types of wear indications.

2. **Industry has taken a similar position with respect to through-wall pitting defects in the EPRI *In Situ Pressure Test Guidelines*, Appendix B, Section B.6. In this section, NUREG/CR 5117 is cited as providing test data from the Surry Unit 2 steam generators as confirmation regarding the burst resistance of deep pit-like defects. Leakage rates above the accident leakage performance criteria for pressure conditions in excess of the accident conditions (up to 3NODP) are not considered consequential with respect to the agreed to structural and leakage integrity performance criteria.**

3. **Finally, any additional requirements regarding allowable leak rate above MSLB pressures up to 3NODP, and the implications of such leakage with respect to severe accident risk is considered to be in excess of currently agreed to deterministic performance criteria.**

In summary, industry does not regard this item to impact the basis for extended inspection intervals. The information contained in NEI 97-06, Integrity Element EPRI Guidelines and the EPRI Flaw Handbook (with regard to the burst and leakage analysis of volumetric flaws) are consistent with the agreed upon burst definition and steam generator performance criteria.

7 *Inspection intervals extending over multiple fuel cycles should be preceded and followed by inspections which utilize qualified NDE techniques for all potential*

degradation mechanisms and locations. Axial SCC is a potential degradation mechanism over the entire tube length. Circumferential SCC is a potential degradation mechanism at locations of geometry variations with length, including expansion transitions, u-bends, and dings or dents.

Industry agrees that qualified NDE techniques should be used. Section 3.1 of Rev. 6 of the PWR SG Examination Guidelines requires that all examinations be conducted with qualified techniques selected in accordance with the degradation assessment. Qualified technique requirements are described in Section 6 of the PWR SG Examination Guidelines.

8. *Indications shall be considered service induced flaw indications in the absence of compelling evidence that the indications are actually associated with manufacturing flaws, surface deposits, tube and/or tube geometry variations, or other inspection artifacts for purposes of determining whether there is active degradation.*

The signal analysis process in the SG Examination Guidelines is intended to be conservative and is sufficient to determine if there are active damage mechanisms. The guidelines require that each of the signals encountered during a steam generator examination be recognized and correctly classified. Also, all crack like indications are considered active damage mechanisms in accordance with the definition For example, Sections 3.3.10 and 3.3.15 of Rev. 6 of the Guidelines stipulate that damage mechanisms not associated with cracking may be experienced in steam generators with Alloy 600TT and Alloy 690TT tubing. If any damage mechanism(s) not associated with cracking is determined to be an active damage mechanism, examination scope shall be expanded or a critical area shall be defined per the requirements of Section 3. Examination periodicity requirements of Sections 3.3.5, 3.4,3.5, and 3.6 apply for these damage mechanism(s) within the critical area. Examination of the critical area may be returned to the examination periodicity requirements of Sections 3.3.10 or 3.3.15 when two examinations have demonstrated that these damage mechanisms are no longer active damage mechanisms, or the operational assessment provides a basis for this return. If an active damage mechanism associated with cracking is present, the tubing shall be subject to the same requirements as Section 3.3.5 for Alloy 600 MA tubing. For damage mechanisms associated with cracking, if subsequent examinations verify active damage mechanisms are not present, the alloy 600 MA requirements still apply. For all damage mechanisms, if evidence (e.g. tube pull, UT, alternate technique(s), historical review of baseline data) proves the damage mechanism was never present, the examination periodicity of Sections 3.3.10 and 3.3.15 may resume.

9. *If primary-to-secondary leakage exceeds 5 gpd prior to shutdown for a refueling outage, an inspection in accordance with the EPRI SG Examination Guidelines for leaker forced outages shall be performed as a minimum.*

The PWR SG Examination Guidelines, Revision 6, Section 3.7, "Primary-to-Secondary Leakage", provides the following guidance with respect to inspections and primary-to-secondary leakage:

"Unexpected low level primary-to-secondary leakage that develops during operation is to be evaluated per the EPRI Primary-to-Secondary Leakage guidelines, EPRI TR-104788 latest revision. If the leakage is $\geq$ 5GPD in any one steam generator, the steps of Section 5.5 shall be followed at the next refueling outage. If the primary-to-secondary leakage assessment performed in accordance with Section 5.5 does not identify the source of the leakage, subsequent assessment during future refueling outage(s) is not required if primary-to-secondary leakage trend is not increasing."

Section 5.5, " Leakage Forced Outage Assessment", defines the inspection requirements for a leakage forced outage. This section states:

> **"This section provides requirements for primary-to-secondary leakage assessment. Degradation detected during an inspection shall be evaluated against structural integrity and accident induced leakage performance criteria. The condition monitoring and operational assessment requirements described in the "EPRI Steam Generator Integrity Assessment Guidelines" shall be performed to provide assurance that the performance criteria will be met until the next scheduled steam generator inspection. In addition, a root cause determination shall be made and included as part of the operational assessment report for leakage forced outages. A leakage forced outage can result from incorrect assumptions or errors in the steam generator program.**
>
> **The steps below shall be followed to establish information about the leak.**
>
> 1. **Determine which steam generator(s) are leaking: Monitor all steam generators to determine which steam generator(s) are leaking.**
>
> 2. **Determine the source of the leakage- This is typically performed by a hydrostatic test, bubble test, or helium leak test to identify suspect tube(s)locations on the tubesheet. Quantify the rate (e.g. drops per minute, gallons per minute.) of leakage. Correlate the calculated**

leakage (pressure/temperature adjusted leakage) versus the operational leakage. Determine if results have accounted for the observed operational leakage, while recognizing that an accurate comparison of operating and shutdown leakage measurements is difficult. If the source of the leakage cannot be identified using the methods described above, 100% eddy current examination should be considered. If the eddy current examination locates the potential leakage proceed with step 4. If the leakage has not been identified an evaluation of the actions within step 6 should be considered.

3. **Examine leaking location(s): This inspection is typically performed by bobbin coil eddy current examination to establish axial location within the steam generator.**

4. **Examine to determine extent, orientation and morphology: This is typically performed by rotating coil technology.**

5. **Review prior inspection history: Review the information contained in the database and review the actual historical bobbin and rotating data to establish factual information about the data. If the leakage is originating from a plug or sleeve review the installation records for that location. Evaluate if installation parameters were met and identify any inconsistencies or nonconforming conditions.**

6. **Perform a root cause evaluation that includes all steam generator program elements in accordance with the utility's program(s). This root cause should evaluate the need to perform eddy current and/or secondary side visual inspections. Also consider supplementing the root cause team with industry peers. The root cause team shall identify immediate, short term and long term actions to correct any process deficiencies.**

7. **Execute root cause corrective action(s).**

8. **Update and revise the degradation assessment, condition monitoring and operational assessments as necessary to address the unexpected leakage.**

9. **Perform required repairs."**

Requested Information

1. *Provide detailed information on degradation experience with tubes and sleeves fabricated from Alloy 600 TT and 690 TT, both foreign and domestic.*

   See attached report, "Experience of U.S. and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves"

2. *Provide additional information concerning hundreds of reported SCC indications in 600 TT tubing worldwide and discuss whether there is a preponderance of evidence than none of these indications are actually SCC.*

   See attached report, "Experience of U.S. and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves"

3. *Submit revised, complete proposal for prescriptive limits on inspection intervals, including supporting definitions.*

   See Section 3 of the attached Rev. 6 of the PWR SG Examination Guidelines

4. *Submit proposed industry protocol for ensuring that the initial occurrence of SCC, industry wide, for Alloy 600 TT or Alloy 690 TT is communicated to all applicable licensees. This protocol should identify the reference temperature at which the "time to detectable cracking" is determined. This protocol should also address the communication of subsequent findings (industry wide) of crack indications occurring after equivalent, accumulated full power operating times less than that observed earlier.*

   See Section 2.3, Items 25 and 30 of attached Rev. 6 of the PWR SG Examination Guidelines.